PERFORMANCE OF PARR-1 WITH LEU FUEL

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ABSTRACT

Pakistan Research Reactor (PARR-1) went critical in 1965 with HEU fuel. The reactor core was converted to LEU fuel with power upgradation from 5 MW to 10 MW in 1992. The reactor has been operated with LEU fuel for about 10,000 hours and has produced about 66,000 MWh energy up to now. Average burn up of the irradiated fuel is about 42 %. The fuel performance during the last 12 years has been excellent. Post irradiation visual inspection of the fuel has revealed no abnormality. During operation there have been no signs of releases in the pool water establishing the full integrity of this fuel. The reactor has been mainly utilized for radioisotope production, beam tube experiments including neutron diffraction studies, neutron radiography etc. Studies have been completed to operate the reactor with a mixed core (HEU + LEU) to utilize the less burned HEU fuel elements. A major project of production of fission Moly using PARR-1 is in the final stages.

1. Introduction

PARR-1, a swimming pool, MTR type reactor attained full power of 5 MW on June 22, 1966 with 93 % Highly Enriched Uranium (HEU) fuel. The reactor is cooled and moderated by light water. Light water and graphite act as the reflector. Since its commissioning, PARR-1 has been mainly utilized for studies in solid state physics and neutron diffraction, nuclear structures, fission physics, neutron activation analysis (NAA), radioisotope production and training of scientists, engineers and technicians. The reactor was operated with HEU fuel for about 30,000 hours and produced about 93,000 MWh energy.

The reactor was shut down in 1990 for core conversion to commercially available LEU fuel. During the process of core conversion the reactor power was also upgraded to 10 MW to meet the demand of higher neutron flux and to compensate the penalty in neutron flux due to conversion from HEU to LEU fuel. Most of the reactor systems including primary and secondary heat transport systems were renovated and several additional systems were installed. IAEA also provided technical assistance for the completion of this project. PARR-1 went critical with <20% LEU fuel on October 31, 1991 and attained the upgraded power level of 9 MW on May 7, 1992. The reactor power was raised to 10 MW in 1998 after enhancing the primary flow rate.

2. Performance of LEU fuel

The reactor has been operated with LEU fuel for about 10,000 hours and has produced about 66,000 MWh energy up to now. Average burn up of the irradiated fuel is about 42 %. The fuel performance during the last 12 years has been excellent. Post irradiation visual inspection of the fuel has revealed no abnormality. During operation there have been no signs of releases in the pool water establishing the full integrity of this fuel. The LEU equilibrium core consists of 29 standard fuel elements and 5 control fuel elements. The core configuration is shown in Fig. 1.

F	E	D	C	B	A	
						1
FC	WB	GR	FC	GR	GR	2
GR	GR	GR	GR	GR	GR	3
SFE	SFE	SFE	WB	SFE	SFE	4
SFE	SFE	CFE	SFE	SFE	SFE	5
SFE	SFE	SFE	SFE	CFE	SFE	6
SFE	CFE	SFE	WB	SFE	SFE	7
SFE	SFE	SFE	SFE	CFE	SFE	8
SFE	SFE	CFE	SFE	SFE	SFE	9

Thermal Column

Fig. 1 Equilibrium LEU core configuration

SFE	Standard fuel element
CFE	Control fuel element
WB	Water box
FC	Fission chamber
GR	Graphite reflector element

F	E	D	C	B	A	
						1
FC	WB	GR	FC	GR	WB	2
WB	GR	WB	GR	GR	GR	3
LS	HS	HS	WB	HS	LS	4
LS	LS	HC	HS	HS	LS	5
LS	LS	LS	LS	LC	LS	6
LS	LC	LS	WB	LS	LS	7
LS	LS	LS	LS	L C	LS	8
LS	LS	LC	LS	LS	LS	9

Thermal Column

Fig. 2 Mixed LEU+HEU core configuration

LS	Standard fuel element (LEU)
LC	Control fuel element (LEU)
HS	Standard fuel element (HEU)
HC	Control fuel element (HEU)
WB	Water box
FC	Fission chamber
GR	Graphite reflector element

3. Reactor utilization

The reactor has been mainly utilized for radioisotope production, beam tube experiments including neutron diffraction studies, neutron radiography, neutron activation analysis and training of scientists, engineers and technicians etc. During the year 2003-2004, 96.73 Ci radioisotope was produced. The major component of radioisotope produced was of ^{131}I which was supplied to 13 medical centres in the country. Other recipients of the radioisotope are various universities and research organizations in the country.

4. Mixed fuel core (LEU + HEU)

When the core was converted from HEU to LEU, many of the spent HEU elements had not reached their design maximum burn up (30%). Burn up of some elements was as low as ranging from 4 to 21%. In order to save money on the purchase of costly fresh LEU fuel elements, it is being considered to use some of the less burnt HEU spent fuel elements along with the present LEU fuel elements. A study was carried out of a proposed mixed core (Fig. 2). Neutronic [1] and steady state thermal hydraulic [2] analyses of this core have been carried out. Computer codes (WIMSD-4, CITATION, PARET) and standard correlations have been employed to calculate different parameters. The results of these analyses are shown in Tables 1 and 2.

The results were compared with the calculated/measured data for an operational LEU core. The reactivity worth of control rods and the average power density in the fuel regions for mixed core is slightly on higher side while the total power peaking factor for mixed fuel core is about 6% higher. Steady state thermal hydraulic results show that mixed fuel core can be operated at 9.8 MW without compromising on reactor safety. The core will have sufficient margins against onset of nucleate boiling, onset of flow instability and departure from nucleate boiling.

Table 1 Neutronic Parameters for the Proposed PARR-1 Mixed Fuel Core

Condition	Core excess reactivity (pcm)		
CZP (BOL)	5659		
HFP (BOL)	2353		
5-FPD	1979		
10-FPD	1642		
15-FPD	1299		
20-FPD	955		
25-FPD	617		
30-FPD	278		
Condition	**Control rod inserted**		
CZP-BOL	~30.0 cm (50.0%)		
HFP-BOL	~17.0 cm (28.3%)		
Condition	**Power peaking factors for critical conditions**		
	Axial	Radial	Total
CZP-BOL	1.529	2.089	3.193
HFP-BOL	1.408	2.183	3.073
Control rod location	**Control rod worth for CZP-BOL critical condition (pcm)**		
D5	-3169		
B6	-2731		
E7	-3221		
B8	-1785		
D9	-15527		
Sum of all rods worth	-12433		
Integrated worth	-13103		

Table 2 Steady-State Thermal Hydraulics

	LEU equilibrium Core	**Proposed Mixed Fuel Core**
Operating Power (MW)	10	9.8
Overpower trip level (MW)	11.5	11.27
Total flow rate (m^3/h)	950	950
Maximum coolant velocity (m/s)	2.46	2.40
Critical velocity (m/s)	10.5	10.5
Power peaking factors:		
- Axial	1.303	1.408
- Radial	2.228	2.183
- Engineering	1.584	1.584
- Total	4.598	4.868
Steady-state temperatures (°C):		
- Coolant temperature rise across		
. Average channel	9.4	9.15
. Hot channel	33.6	31.47
. Core (including bypass flow)	8.5	8.26
- Peak clad surface temperature	102.47	108.94
- Peak centerline temperature	104.64	111.52
Average heat flux (W/cm^2)	18.1	18.87
Peak heat flux (W/cm^2)	83.4	91.86
Onset of nucleate boiling (ONB)		
- Average heat flux (W/cm^2)	25.52	24.0
- Peak heat flux (W/cm^2)	117.34	116.9
- Location of ONB from top(cm)	44	44
- Peak temperatures (°C):		
. Fuel centerline	128.5	129.2
. Clad surface	125.5	125.9
. Coolant exit	85.4	79.9
Onset of Flow Instability (OFI):		
- Peak heat flux (W/cm^2)		
. Forgan	138	144
.CEA	170	174
Departure from Nucleate Boiling (DNB)		
- Critical heat flux (W/cm^2)		
. Labunstov	326	321.9
. Mirshak	257	253.9

Safety margins:		
- Margin to ONB	1.4	1.3
- Margin to OFI		
.Forgan	1.6	1.6
.CEA	2.0	1.9
- Margin to DNB		
. Labuntsov	3.9	3.5
. Mirshak	3.1	2.8

5. Production of Fission Molybdenum-99 for preparation of Tc-99m generators

^{99m}Tc is a short-lived ($T_{1/2}$ = 6 h) daughter product of the parent Molybdenum-99 ($T_{1/2}$ = 66 h), which is mainly produced by the nuclear fission of uranium-235 (^{235}U). Small amounts of ^{99}Mo are also produced by the neutron activation method. The current applications of ^{99}Mo $\rightarrow$ ^{99m}Tc generators in oncology, cardiology and other fields almost completely depend on the fission production of ^{99}Mo.

More than 20 ^{99m}Tc generators loaded with fission Molybdenum-99 are being consumed weekly in different nuclear medical centers and hospitals in Pakistan. IAEA has already provided ^{99}Mo loading facility to PINSTECH under their Technical Cooperation Programme, in which ^{99m}Tc generators (PAKGEN) conforming to the international standards are being manufactured and supplied to various nuclear medical centers. However, fission Molybdenum-99 loaded in these generators is being imported from South Africa. To overcome the problems associated with import of fission ^{99}Mo such as hard currency, increasing price of ^{99}Mo, import policies, delay and changes in supply schedules, etc. the indigenous production of fission ^{99}Mo in the country has been proposed. The Planning Commission of Pakistan has recently approved a project "Production of Molybdenum-99 for Medical Use" at PINSTECH, Islamabad. A time frame of 2 years is specified for the completion of the project. For transfer of technology, a formal agreement between PAEC and German firm Hans Waelishchmiller GmbH, BT Dresden Germany has been accomplished. Most of the equipment will be imported. The plant will be capable of producing sufficient amount of Molybdenum-99 for domestic utilization as well as for export. The proposed facility will have the capacity to produce 500 Curies of Molybdenum-99 per batch.

The facility will be installed in laboratories near PARR-1 which will allow a safe transfer of irradiated U-235 targets to processing hot cells. A total area of about 320 m^2 will be used. The processing facility will be installed in the main hall (17.9m x 10.5m), while adjacent laboratories will be used as health physics post, target preparation, storage, quality control, interim storage of waste, etc.

The building/laboratories will be renovated/constructed and fitted with special ventilation and exhaust systems required for radiation work as per recommendations of IAEA. The Molybdenum-99 separation process will be installed in 3 hot cells which will be equipped with master slave manipulators, lead glass windows and stainless steel lining. These will also be equipped with waste outlet devices at the bottom of the cells, gas inlets and conveyer system for transfer of materials. The exhaust of all the cells will be connected through a radioiodine filter as well as particle filter before being fed to the general ventilation system. For the irradiation of uranium-235 in reactor, the target has been designed and fabricated in the same manner as fuel for nuclear reactors. Self-propelled shielded containers for irradiated targets would be designed for radiation safety reasons. Rigorous quality control procedures will be adopted to monitor the purity of Molybdenum-99 for its use in nuclear medicine. Radioactive wastes generated during the separation of fission Molybdenum-99 from neutron-irradiated uranium will be treated and disposed off in environmentally acceptable ways.

For PARR-1 with a flux of $1x10^{14}$ n/cm^2-sec, the yield of ^{99}Mo per gram U-235 is shown in the following table against the irradiation time[3]:

Irradiation Time	^{99}Mo Yield
8 hr	19.40 Ci
12 hr	28.51 Ci
16 hr	37.24 Ci
20 hr	45.62 Ci
24 hr	53.65 Ci
48 hr	95.35 Ci
168 hr	199.57 Ci

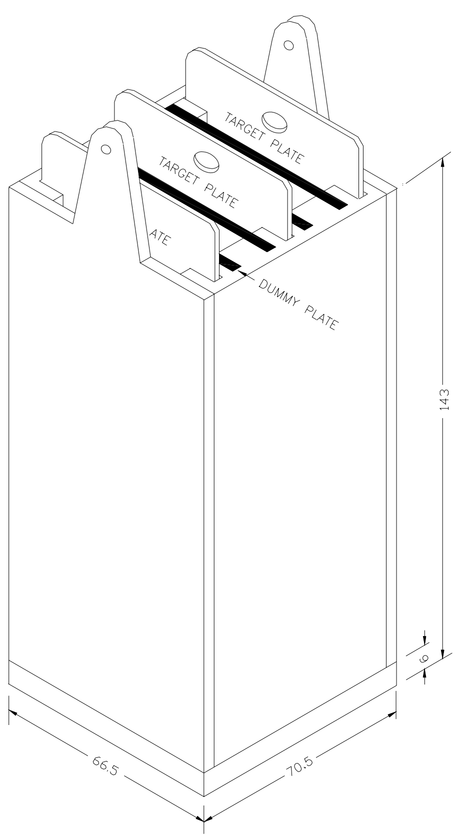

Fig.3 Target holder for U-235 irradiation at PARR-1

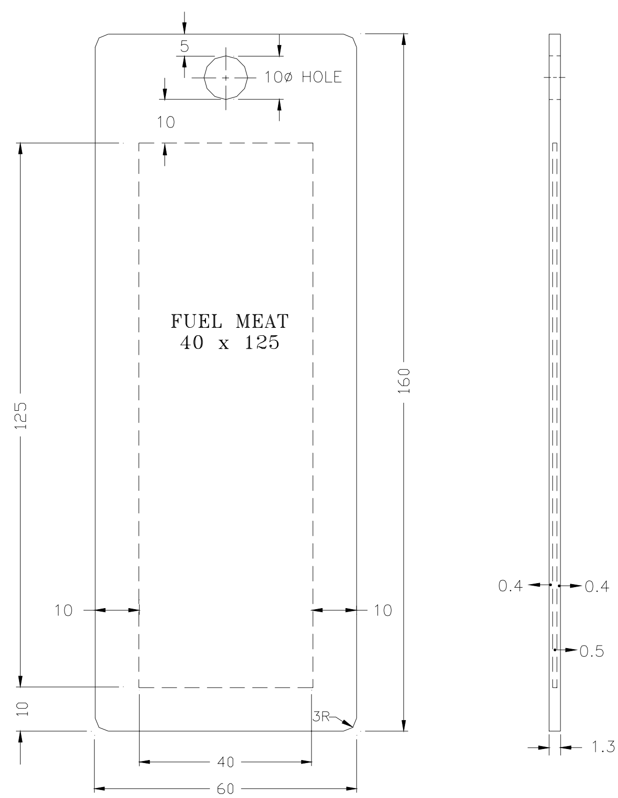

Fig. 4 U-235 target plate

Demand at reference date = 12 Ci
Time elapsed during;

(i)	Cooling of target and chemical separation of ^{99}Mo	=	2 days
(ii)	Preparation of ^{99m}Tc generators	=	2 days
(iii)	Transportation of ^{99m}Tc generators	=	1 day
(iv)	Arrival at hospital (early)	=	2 days
(v)	Total time elapsed after irradiation	=	7 days

Hence Reactor yield should be 70 Ci
Considering chemical processing yield = ~ 73%
The reactor yield should be 97 Ci.

Table 3 Relationship between target material, irradiation time and ^{99}Mo yield.

^{235}U (grams)	Neutron Flux	Irradiation Time	^{99}Mo yield at E.O.I.
1	$1x10^{14}$ n cm^{-2} s^{-1}	8 hour	19.4 Ci
5	$1x10^{14}$ n cm^{-2} s^{-1}	8 hour	97.0 Ci
3.4	$1x10^{14}$ n cm^{-2} s^{-1}	12 hour	97.0 Ci
2.604	$1x10^{14}$ n cm^{-2} s^{-1}	16 hour	97.0 Ci
2.126	$1x10^{14}$ n cm^{-2} s^{-1}	20 hour	97.0 Ci
1.807	$1x10^{14}$ n cm^{-2} s^{-1}	24 hour	97.0 Ci
1.017	$1x10^{14}$ n cm^{-2} s^{-1}	48 hour	97.0 Ci

6. Results and Conclusions

- PARR-1 has been safely operated with LEU fuel for about 12 years and the fuel has proved to be integral and safe.
- To make use of the less burned HEU fuel which was in use before conversion to LEU fuel, a mixed LEU/HEU core is suggested and has been analyzed to be safe.
- The reactor is currently being utilized for radioisotope production and beam tube experiments.
- In view of country’s need and interest, a plan for construction of a facility for production of fission ^{99}Mo is underway which will lead to more reactor utilization.

7. References

[1] M. Arshad, A. Manan, & M. Sagheer, “Neutronic Calculations for Mixed PARR-1 Equilibrium Core Using Highly Enriched and Low Enriched Fuel Elements”, technical report INUP-TR/172, Institute of Nuclear Power, Nilore, Islamabad, Pakistan, Nov 2003

[2] I.H. Bokhari, “Steady-state Thermal Hydraulic and Safety Analyses of a Proposed Mixed Fuel (HEU & LEU) for Pakistan Research Reactor-1”, Annals of Nuclear Energy, Vol 31, Issue 11, pp. 1265-1273, 2004

[3] M. Jehangir & A. Mushtaq, “Production of Fission Molybdenum-99 for Preparation of Technetium-99m Generators for Medical Use”, technical report IPD-REP-3, Pakistan Institute of Nuclear Science and Technology, Islamabad, Pakistan, January 2003